NON-PUBLIC?: N
ACCESSION #: 8902240116
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000414

TITLE: Reactor Trip Following Failure of a Main Feedwater Control Valve Fuse
Due To Manufacturing and Management Deficiencies
EVENT DATE: 01/12/89 LER #: 89-001-00 REPORT DATE: 02/10/89

OPERATING MODE: 1 POWER LEVEL: 094

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv) and OTHER: 10CFR50.72(b)(2)(ii)

LICENSEE CONTACT FOR THIS LER:
NAME: Julio G. Torre,
Associate Engineer - Licensing TELEPHONE: (704) 373-8029

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SJ COMPONENT: FV MANUFACTURER: Bussmann
REPORTABLE TO NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On January 12, 1989, at 0943 hours, a Bussmann FNA fuse failed in a Main Feedwater Control Valve circuit causing the valve to fail closed. Control Room Operators opened the Main Feedwater Bypass Valve in an attempt to provide additional feedwater. The unexpected closure isolated feedwater flow to Steam Generator 2D and quickly resulted in an automatic Reactor trip. Unit 2 was in Mode 1, Power Operation, at 94% power when this incident occurred. Previous mechanical failures of Bussmann FNA fuses in 1986 at McGuire and Catawba prompted Design Engineering to identify suitable replacements for all Class 1E applications. Scheduling of non-safety FNA fuse replacement was decided to be left to the discretion of the station as that was considered to be a reliability issue.

This incident has been attributed to a manufacturing deficiency. The fuse which closed the Main Feedwater Control Valve failed mechanically. This incident has also been attributed to a management deficiency. Although the mechanical failures on the Bussmann FNS fuse had been previously identified,

non-safety applications and suitable replacements were not implemented in a timely manner. Following this incident, all Bussmann FNA fuses on both Units' Main Feedwater Control and Bypass Control valves were replaced with Littelfuse FLQ fuses. Also, plans were developed to change out all remaining Bussmann FNS fuses. The health and safety of the public were unaffected by this event.

END OF ABSTRACT

The Main Feedwater EIIS:SJ! (CF) System supplies feedwater to the four Steam Generators EIIS:SG! (S/Gs) to maintain proper S/G water levels that correspond with Reactor EIIS:RCT! power output and Turbine EIIS:TRN! steam requirements. Each of the four CF lines contains a CF Control Valve EIIS:V!: 1/2 CF28, 37, 46, and 55) and a CF Bypass Control Valve (1/2 CF30, 39, 48, and 57). The CF Control Valves and CF Bypass Control Valves are normally modulated by the CF Control System to maintain proper S/G water inventory. The CF Bypass Control Valves control feedwater flow up to 15% load and the CF Control Valves handle flow from 15% to 100%. These valves close on CF Isolation, or loss of control power.

The Bussmann FNA fuse is EIIS:FU! a single element indicating fuse. There have been numerous mechanical failures with these fuses. The failure point is the interface between the short circuit element to the solder which connects the short circuit element and the overload element. The Littelfuse FLQ fuse is a non-indicating single element fuse which is recommended by Design Engineering to replace the Bussmann FNA fuse in both mild and harsh safety related environments.

DESCRIPTION OF INCIDENT:

On January 12, 1989, Unit 2 was operating in Mode 1, Power Operation, at 94% power. One of the four Bussmann FNA fuses for 2CF55, S/G 2D CF Control Valve mechanically failed causing the valve to close at 0942:40 hours. The Control Room received a S/G 2D Flow Mismatch Lo CF Flow alarm. Control Room Operators (CROs) verified full demand to 2CF55. The CRO opened 2CF57, S/G 2D Bypass Control Valve, in an attempt to increase feedwater flow. At 0943:10:273 hours, S/G 2D Low Low Level Reactor Trip alarmed in one out of four S/Gs. The Reactor tripped on low low level in S/G 2D. Reactor Coolant EIIS:AB! Pumps EIIS:P! 2C and 2D #1 seal EIIS:SEL! indicated leakoff flow went to zero gpm and remained there for 19 and 11 minutes, respectively. The Reactor trip also caused a Turbine trip, as expected.

Both Motor Driven Auxiliary Feedwater EIIS-BA! (CA) Pumps autostarted on one out of four S/Gs in low low level. The Turbine Driven CA Pump (CAPT) subsequently started due to S/G 2C and 2D being in low low level. Main

Feedwater Isolation occurred as expected, as a result of Reactor trip with low Tavg. All appropriate Nuclear Sampling EIIS:CB! (NM) and S/G Blowdown EIIS:WI! (BB) valves closed as expected, except 2NM190A, S/G 2A Blowdown Sampling EIIS:CB! Containment Isolation. An Operator was dispatched to manually close the valve. Operations Work Request 42592 OPS was written to investigate/repair 2NM190A. All Feedwater Isolation valves indicated closed as expected, except 2CF28, S/G 2A CF Control Valve, which was later verified as closed.

The CRO reset the CA autostart signal. At 0946:46 hours, the CRO secured the CAPT. CA was throttled by the CROs to limit Reactor Coolant EIIS:AB! (NC) System cooldown. Condenser EIIS:COND! Steam Dump (SB) valves opened as expected, to control Tavg. However, 2SB-15, SM Bypass to Condenser Control Valve, did not reclose. All S/G Low Low Narrow Range Level alarms subsequently cleared by 1027 hours. The CRO completed realignment of NM and BB valves by 1131 hours. The CRO completed realignment of CF valves by 1459 hours.

On January 13, 1989, Unit 2 Main Feedwater Control and Bypass Control Valve Bussmann FNA fuses were replaced with Littelfuse FLQ fuses. On January 14, 1989, Unit 1 Main Feedwater Control and Bypass Control Valve Bussmann FNA fuses were replaced with Littelfuse FLQ fuses.

On January 14, at 1448 hours, Unit 2 entered Mode 2, Startup. At 1820 hours, Unit 2 entered Mode 1, Power Operation.

CONCLUSION:

This incident has been attributed to a manufacturing deficiency. The Bussmann FNA-2A fuse (manufacturer No. 782112) which isolated Feedwater flow to S/G 2D failed mechanically. Previous failures of Bussmann FNA fuses prompted inspections, and a Design Study to identify suitable replacements for safety related harsh environment applications. The study was completed in October 1988, and changeouts of safety related Bussmann FNA fuses in harsh environments were planned to occur after the Refueling Outages on both Units which were expected to be completed by April 1989. Non-safety FNA replacement would have likely occurred subsequently.

This incident has also been attributed to a management deficiency. The Bussmann FNA fuses had been identified as having an unacceptable mechanical failure probability in June 1986. The identification of a suitable replacement for harsh environment safety related applications was not completed until October 1988 due to testing requirements. The fuse problem was initially identified in a Duke Power Non-Conforming Item Report (NCI) in

July 1986. The safety related mild environment replacement was completed in November 1987. Since the harsh environment changeout was still outstanding, the NCI was converted to a Duke Power Problem Item Report (PIR) in August 1987. Ongoing resolution continued with the issuance of a Design Study in October 1988 which dealt with safety related Bussmann FNA fuses located in harsh environments. It should be noted that NCI CN-438 was written on mechanical failures of all Bussmann FNA fuses. Also, NRC IE Information Notice 87-62 was issued in December 1987 describing similar fuse failures. After NCI CN-438 was rewritten as PIR 0-C87-0232, the resolution (Duke Power Design Study 064) was directed at only safety related circuits. The need to replace all FNA fuses was initially recognized by Design Engineering and Nuclear Production. In prioritizing the replacement effort, non-safety related FNA fuse replacement was considered to be a reliability issue and was left to the discretion of the station for scheduling in light of existing outage schedules.

The control of identified problems was consolidated in October 1986 under the PIR program. Prior to that, problems were identified under various programs& such as NCIs and Station Problem Reports. Resolution of these fuse problems should have occurred in a more timely manner. The replacement of non-safety FNA fuses was not identified as a corrective action in either the original NCI Program or the PIR Program. All groups involved in developing or reviewing resolutions should be made aware of the need to address and identify the full scope of all problem resolutions.

A review of NPRDS on Main Feedwater valve failures due to mechanical fuse failures reveals four previous events, two of which were due to the same Bussmann FNA fuse at McGuire on March 25, 1986, and Catawba on October 23, 1987.

A review of the Operating Experience Program database shows five previous incident investigation reports due to fuse problems. Only one of these five was due to a failed fuse, that being a Chase-Shawmutt fuse in the Rod Control EIIS:AA) System (Ler 414/88-22). The Mechanical fuse failures are considered to be a recurring problem. The corrective action taken for these previous events would not have prevented this Bussmann FNA fuse failure.

Failure of the Bussmann FNA fuse is not specifically NPRDS reportable, but the unexpected closure of 2CF55 will be reported.

CORRECTIVE ACTION:

SUBSEQUENT

(1) Unit 2 S/G Main Feedwater Control and Bypass Control valves Bussmann FNA fuses were replaced on January 13, 1989.

(2) Unit 1 S/G Main Feedwater Control and Bypass Control valves Bussmann FNA fuses were replaced on January 14, 1989.

(3) Standing Work Requests 7036 and 7037 are being performed weekly to inspect safety related Bussmann FNA fuses that are in harsh environments.

PLANNED:

(1) Completion of safety related Bussmann FNS fuse changeout on both Units will be identified.

(2) Responsibility for resolution of all inadequate post-trip response will be identified and all items will be identified on station commitment index.

(3) 2SB-15 will be investigated/repaired under Work Request 41984 OPS.

(4) A listing by cabinet and fuse location for all non-safety related FNS fuses at Catawba will be provided.

(5) Completion of changeout of all Bussmann FNS fuses on both units will be identified.

(6) All appropriate groups involved in developing or reviewing problem resolutions will be notified to be aware of the need to address and identify the full scope of all problem resolutions.

SAFETY ANALYSIS:

The Reactor trip was automatically initiated from 94% full power due to S/G 2D low-low level. The Reactor Trip Breakers EIIS:BRK! opened within 66 milliseconds of the Solid State Protection EIIS:JC! System Trip signal, and all of the control rods inserted into the core, reducing power to decay heat level. Feedwater Isolation was automatically initiated upon Reactor trip with low Tave (564 degrees F). Both Motor EIIS:MO! Driven Auxiliary Feedwater (CA) Pumps were autostarted upon Steam Generator (S/G) 2D low-low level, and a Turbine Driven CA Pump Autostart signal occurred approximately 11 seconds later upon low-low level in two-out-of-four S/Gs. The redundant steam supply valves for the Turbine Driven CA Pump, 2SA2 and 2SA5, opened within 4 seconds of the autostart signal.

Immediately prior to the trip, Reactor Coolant (NC) Loop 2D temperature increased 2 degrees F upon 2CF55 closure, to a maximum of 592 degrees F. Upon Reactor Trip, temperature in all four loops decreased to a minimum of 550 degrees F, and then stabilized at 555 degrees F within 30 minutes post trip, 2 degrees F from the no-load target of 557 degrees F. The NC System pressure increased 30 psig upon closure of 2CF55, to a maximum value of approximately 2258 psig. Upon Reactor Trip, NC pressure decreased to a minimum value of approximately 1961 psig, and then stabilized at 2240 psig within 40 minutes post-trip, 5 psi from the no-load target of 2235 psig. Pressurizer EIIS:PRZ) level increased 1% upon 2CF55 closure, to a maximum value of 60%. Upon Reactor Trip, pressurizer level decreased to a minimum value of 23%, and stabilized at the no-load target of 25% within 40 minutes post-trip. Steam pressure increased to a maximum value of 1098 psig upon Turbine trip, and then decreased to a minimum of 992 psig due to steam dump to condenser and the quenching effect of CA flow. Within 40 minutes post-trip, steam pressure stabilized at 1056 psig, 34 psi from the no-load target of 1090 psig. S/Gs 2A, 2B, 2C, and 2D reached a minimum wide range indicated value of 46%, 45%, 47%, and 30%, respectively. Correction of these values for calibration condition variations yields actual wide range minimum levels of 60%, 59%, 61%, and 37% for S/Gs A, B, C, and D, respectively.

Immediately post-trip, the Operators entered the Reactor Trip or Safety Injection Emergency Procedure. The Operators secured the Turbine Driven CA Pump (approximately 3.5 minutes after the trip), throttled CA flow, and closed drain valves off the Main Steam equalization header to prevent excessive primary side cooldown. The Operators assumed manual control of pressurizer level to maintain level and prevent automatic letdown isolation.

The S/G PORVs, Coded Safety valves, and Atmospheric Dump valves remained closed post-trip, and all three banks of Steam Dump to Condenser valves opened post-trip to dump steam to the condenser. As required by the Reactor Trip Emergency Procedure, the Operators maintained CA flow above the minimum required cumulative value of 450 gpm to the S/Gs while S/G level was less than 47% indicated wide range. The Reactor Coolant was 45 degrees F subcooled at the point of minimum Reactor Coolant System pressure. Adequate heat sink for core decay heat removal was available and maintained at all times.

Section 15.2.7 of the Catawba FSAR assumes that a) the Reactor is tripped at 102% of ESF design rated power, b) the CA System is autostarted one minute after the S/G Low-Low Level signal, and c) steam is relieved through the S/G PORVs and Coded Safety valves. In this event, the Unit was tripped from 94% full power. Since offsite power was available, the Motor Driven CA Pumps were autostarted when the low-low level occurred. Also, while the S/G PORVs and

Coded Safety Valves were available for use if needed, the Steam Dump to Condenser valves were used to dump steam and dissipate core residual heat. Therefore, this event is fully bound by the "Loss of Normal Feedwater Flow" transient as described in Section 15.2.7 of the Catawba FSAR.

All plant safety equipment was available throughout this event. The cooldown limits of 100 degrees F per hour for the Reactor Coolant System and 200 degrees F per hour for the pressurizer were not exceeded. Integrity of the fuel cladding, Reactor Coolant System, and Containment structure was maintained at all times.

During the post-trip plant response, 2CF28, S/G 2A CF Control Valve, was observed by Control Room indicating lights to travel to an intermediate position. Based on repair work performed later, it is likely that mechanical problems prevented 2CF28 from stroking completely closed when CF Isolation was initiated. 2CF28 is not a Containment Isolation valve, It is a Feedwater Isolation valve, and its function is to terminate additional heat and mass input to Containment in certain FSAR Chapter 15 scenarios. However, 2CF33, S/G 2A CF Containment Isolation Valve (which closes upon CF Isolation), automatically closed for termination of CF flow and isolation of the header to S/G 2A. 2CA149, S/G 2A CF Bypass to CA Nozzle Valve, automatically closed for termination of flow and isolation of the CF Bypass to CA Header for S/G 2A. Therefore, the CF Isolation design function was accomplished during this event.

This incident is reportable pursuant to 10CFR 50.73, Section (a)(2)iv) and 10CFR 50.72, Section s(b)(2)(ii).

The health and safety of the public were not affected by this incident.

ATTACHMENT 1 TO 8902240116 PAGE 1 OF 1

Duke Power Company HAL B. TUCKER
P.O. Box 33198 Vice President
Charlotte, N.C. 28242 Nuclear Production
(704)373-4531

CITIZENSHIP
DUKE POWER DUKE POWER
SERVICE

February 10, 1989

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/89- 01

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/89-01 concerning a reactor trip following the failure of a Main Feedwater Control Valve fuse due to manufacturing and management deficiencies. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

JGT/IIR88

xc: M. L. Ernst, Acting American Nuclear Insurers
Regional Administrator, Region II c/o Dottie Sherman, ANI Library
U. S. Nuclear Regulatory Commission The Exchange, Suite 245
101 Marietta Street, NW, Suite 2900 270 Farmington Avenue
Atlanta, Georgia 30323 Farmington, CT 06032

M & M Nuclear Consultants Mr. W. T. Orders
1221 Avenue of the Americas NRC Resident Inspector
New York, New York 10020 Catawba Nuclear Station

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia

*** END OF DOCUMENT ***